April 7, 1997

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: ISSUANCE OF AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M97790)

Dear Mr. Boulette:

The Commission has issued the enclosed Amendment No. 171 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated January 24, 1997, as supplemented March 27, 1997.

This amendment revises Technical Specifications Safety Limit 2.1.2, "Minimum Critical Power Ratio" and the associated Bases section and Note 5 to Table 3.2.C.1, "Instrumentation that Initiates Rod Blocks." This change is necessary as the generic safety limit minimum critical power ratio (SLMCPR) calculation for the Pilgrim Plant was non-conservative when plant cycle specific parameters are used in the SLMCPR analysis.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

(Original Signed By)

Alan B. Wang, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 171 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

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DATED: April 7, 1997

AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-35-PILGRIM NUCLEAR POWER STATION

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ACRS
PD plant-specific file
R. Conte, Region I

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 7, 1997

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Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
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Plymouth, MA 02360

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Office of Nuclear Reactor Regulation

Docket No. 50-293

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2. Safety Evaluation

cc w/encls: See next page

cc:

Mr. Leon J. Olivier
Vice President of Nuclear
Operations & Station Director
Pilgrim Nuclear Power Station
1FD #1 Rocky Hill Road
Plymouth, MA 02360

Resident Inspector
U. S. Nuclear Regulatory Commission
Pilgrim Nuclear Power Station
Post Office Box 867
Plymouth, MA 02360

Chairman, Board of Selectmen
11 Lincoln Street
Plymouth, MA 02360

Chairman, Duxbury Board of Selectmen
Town Hall
878 Tremont Street
Duxbury, MA 02332

Office of the Commissioner
Massachusetts Department of
Environmental Protection
One Winter Street
Boston, MA 02108

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, MA 02108

Mr. Robert M. Hallisey, Director
Radiation Control Program
Massachusetts Department of
Public Health
305 South Street
Boston, MA 02130

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Ms. Jane Fleming
8 Oceanwood Drive
Duxbury, MA 0233

Mr. Jeffery Keene
Licensing Division Manager
Boston Edison Company
600 Rocky Hill Road
Plymouth, MA 02360-5599

Ms. Nancy Desmond
Manager, Reg. Affairs Dept.
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

Mr. David F. Tarantino
Nuclear Information Manager
Pilgrim Nuclear Power Station
RFD #1, Rocky Hill Road
Plymouth, MA 02360

Ms. Kathleen M. O'Toole
Secretary of Public Safety
Executive Office of Public Safety
One Ashburton Place
Boston, MA 02108

Mr. Peter LaPorte, Director
Attn: James Muckerheide
Massachusetts Emergency Management
Agency
400 Worcester Road
P.O. Box 1496
Framingham, MA 01701-0317

Chairman, Citizens Urging
Responsible Energy
P.O. Box 2621
Duxbury, MA 02331

Citizens at Risk
P.O. Box 3803
Plymouth, MA 02361

W.S. Stowe, Esquire
Boston Edison Company
800 Boylston St., 36th Floor
Boston, MA 02199

cc: (cont.)

Chairman
Nuclear Matters Committee
Town Hall
11 Lincoln Street
Plymouth, MA 02360

Mr. William D. Meinert
Nuclear Engineer
Massachusetts Municipal Wholesale
Electric Company
P.O. Box 426
Ludlow, MA 01056-0426

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated January 24, 1997, as supplemented March 27, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-35 is hereby amended to read as follows:

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Patrick D. Milano, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
2.1	2.1
B2-1	B2-1
B2-2	B2-2
B2-3	B2-3
B2-4	B2-4
3/4.2-22	3/4.2-22

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be < 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.08.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met the following actions shall be met:

2.2.1 Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.

2.2.2 Within two hours:

A. Restore compliance with all Safety Limits, and

B. Insert all insertable control rods.

2.2.3 The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.

2.2.4 A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.

2.2.5 Critical operation of the unit shall not be resumed until authorized by the Commission.

2.0 SAFETY LIMITS

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish a Safety Limit such that the Minimum Critical Power Ratio (MCPR) is not less than the limit specified in Specification 2.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling (i.e., MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

FUEL CLADDING INTEGRITY (2.1.1)

GE critical power correlations are applicable for all critical power calculations at pressures at or above 785 psig or core flows at or above 10% of rated flow. For operation at low pressures and low flows another basis is used as follows:

(Cont)

FUEL CLADDING INTEGRITY (2.1.1) (Cont)

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

MINIMUM CRITICAL POWER RATIO (2.1.2)

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (2), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1300 psia
Mass Flux:	0.1 to 1.5 Mlb/hr-ft^2
Inlet Subcooling:	0 to 70 Btu/lb
Axial Profile:	1.5 chopped cosine 1.7 inlet peaked 1.7 outlet peaked
R-Factor	0.95 to 1.20
Rod Array	9X9 GE 11 array

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at

(Cont)

MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont)

which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. Reference 1 includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.

REACTOR WATER LEVEL (Shutdown Condition) (2.1.3)

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

(Cont)

INTRODUCTION

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Rod Array	9X9 GE 11 array

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(Cont)

MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont)

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(Cont)

REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.1.0 for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

REFERENCES

1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (Applicable Amendment specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

NOTES FOR TABLE 3.2.C-1

1. With the number of operable channels:

   a. One less than required by the minimum operable channels per trip function requirement, restore an inoperable channel to operable status within 7 days or place an inoperable channel in the tripped condition within the next hour.

   b. Two or more less than required by the minimum operable channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. a. With one RBM Channel inoperable:

   (1) restore the inoperable RBM channel to operable status within 24 hours; otherwise place one rod block monitor channel in the tripped condition within the next hour, and;

   (2) prior to control rod withdrawal, perform an instrument function test of the operable RBM channel.

   b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

3. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

4. SRM operability requirements during core alterations are given in Technical Specification 3.10.

5. RBM operability is required in the run mode in the presence of a limiting rod pattern with reactor power greater than the RBM low power setpoint (LPSP). A limiting rod pattern exists when:

   MCPR < 1.41 for reactor power $\geq$ 90%

   MCPR < 1.72 for reactor power < 90%

   The allowable value for the LPSP is $\leq$ 29% of rated core thermal power.

6. When the reactor mode switch is in the Refuel position, the reactor vessel head is removed, and control rods are inserted in all core cells containing one or more fuel assemblies, these Rod Block functions are not required.

7. With one or more Reactor Mode Switch - Shutdown Position channels inoperable, suspend control rod withdrawal and initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies immediately.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated January 24, 1997, as supplemented March 27, 1997, the Boston Edison Company (BECo) (the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station (PNPS) Technical Specifications (TSs). The requested changes would revise (1) the Safety Limit Minimum Critical Power Ratio (SLMCPR) based on the cycle-specific analysis of the mixed core of GE11/GE10/GE8B fuel parameters and (2) an increase in the minimum critical power ratio (MCPR) criteria that define a limiting control rod pattern required by an increase in the SLMCPR. The March 27, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee requested a change to the PNPS Cycle 12 TSs in accordance with the 10 CFR 50.90. The proposed revision of the TS 2.1.2 and its associated Bases Sections 2.1, 2.1.1, 2.1.2, and Note 5 to TS Table 3.2.C.1 is described below.

(I) TS 2.1.2 and ASSOCIATED BASES SECTIONS

The safety limit MCPR in TS 2.1.2 is proposed to change from 1.07 to 1.08 with the reactor steam dome pressure $\geq$ 785 psig and core flow $\geq$ 10% of rated flow based on the cycle-specific analysis performed by GE for PNPS Cycle 12 mixed core of GE11/GE10/GE8B fuel. The cycle-specific parameters were used including the actual core loading, actual bundle parameters evaluated at the projected exposure distribution based on projected control blade patterns for the rodded burn through the cycle, and the full cycle exposure range to determine the most limiting point(s).

The staff has reviewed the proposed TS change which is based on the analyses performed using PNPS Cycle 12 cycle-specific inputs and approved methodologies including GESTAR II (NEDE-24011-P-A-11, Sections 1.1.5 and 1.2.5) and NEDO-10985-A, January 1977. Because the R-factor methodology referenced in NEDE-24011-P-A-11 is not applicable to the part length

GE11, an improved R-factor methodology described in NEDC-32505P, " R-Factor Calculation Method for GE11, GE12 and GE13 Fuel", November 1995 was used. The revised R-factor calculation method uses the same NRC approved equation stated in GESTAR (NEDE-24011-P-A) with the correction factors to account for the peaking factor effects due to the part-length-rod design. The staff has reviewed the R-factor calculation method for the GE11, the relevant information provided in the proposed Amendment 25 to GESTAR II, NEDE-24011 (which is under the staff review) and the supplemental information dated March 26, 1997 in response to the staff request for additional information during a telephone conference on March 13, 1997. The staff has found that the justification for analyzing and determining the SLMCPR of 1.08 is acceptable for application to the GE11 fuel in PNPS Cycle 12 since Pilgrim Cycle 12 bundle has a flatter distribution of uncontrolled R-factors for the highest power rods in each bundle.

The licensee has revised the associated Bases sections related to the above TS changes for the SLMCPR. The proposed changes modify the Bases section to describe the methodology used in the calculation of the SLMCPR. The new Bases sections have been changed to reflect the new requirements and are consistent with the proposed amendment. The new Bases pages 2.1, 2.2, 2.3, and 2.4 are included with the new TS pages. These changes are acceptable.

(2) Note 5 of Table 3.2.C.1

The staff discussed with the licensee by telecon the proposed SLMCPR change and corresponding changes to MCPR criteria in TS section 3.2.C.1. TS section 3.2.C.1 defines a limiting rod pattern criteria as MCPR < 1.40 for Core Thermal Power greater than 90% and MCPR < 1.70 for Core Thermal Power less than or equal to 90%. These criteria were determined in the ARTS program analysis for Pilgrim. The SLMCPR assumed in this analysis is 1.07. An increase in the SLMCPR will require an increase in the MCPR criteria used to define a limiting rod pattern. For a SLMCPR of 1.08, the MCPR criteria for a limiting rod pattern are MCPR < 1.41 for Core Thermal Power greater than 90% and MCPR < 1.72 for Core Thermal Power less than or equal to 90%. The rod withdrawal error analysis for Pilgrim Station is presented in the "ARTS Improvement Program Analysis for Pilgrim Nuclear Power Station," NEDC-31312-P, September 1987. Applicability of this analysis for more recent fuel designs is verified as part of the reload core design analysis and is documented in the cycle-specific supplemental reload licensing report. Therefore, this change is acceptable to the staff.

The licensee also stated that the linear heat generation rate (LHGR) limit has been reviewed for Cycle 12 for the rod withdrawal error, and that the current operability statement for the rod block monitor utilizing the revised criteria for a limiting rod pattern will provide adequate protection against exceeding the LHGR thermal limit, as well as SLMCPR, during a rod withdrawal error event. The licensee will continue to review the LHGR limit in future cycles. Based on NRC staff review, we conclude that the changes to the TS for Pilgrim Cycle 12 application are analyzed based on the NRC-approved method.

The staff reviewed the request by BECo to revise the TS of the PNPS and based on the review, conclude that these revisions are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts State Official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 6568). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: T. Huang
G. Golub

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